APPENDIX F

ATMOSPHERIC DISPERSION AND DOSE CALCULATIONS FOR NORMAL AND ACCIDENT CONDITIONS

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F. ATMOSPHERIC DISPERSION AND DOSE CALCULATIONS FOR NORMAL AND ACCIDENT CONDITIONS

F.1 INTRODUCTION

This appendix describes the data, methods, and assumptions used to estimate dose to workers and to the public from emissions of radioactive and toxic materials from the SNS. The steps in estimating dose are as follows:

- Identify and quantify emissions (source terms),
- Identify and select human exposure pathways,
- Analyze transport of contaminants through each exposure pathway, and
- Calculate dose.

This sequence of steps was repeated several times as new or more realistic data became available and assumptions refined. The purpose of these dose calculations is to provide reasonable but conservative dose estimates that allow impacts of the alternative actions analyzed in the EIS to be compared.

The radionuclides that would be discharged into the environment by the SNS would be produced in spallation reactions initiated by the high-energy protons generated in the linac. These reactions occur in cascades or "stars" as fragments and neutrons from atomic nuclei struck by high-energy protons strike and react with other atoms until the energy of the initial collision is dissipated. The spectrum of radionuclides and the number of neutrons produced by spallation depend on the energy and intensity of the proton beam and the nature of the material it strikes.

The purpose of the mercury target is to generate neutrons by spallation. The radionuclides formed directly by spallation and by reactions with the neutrons in the target and surrounding materials are waste products. A small fraction of the particles in the beam would also escape from the confining magnetic fields and induce spallation reactions in the components and structures in the linac, beam storage, beam transfer tunnels, in the beam stops, and in the target areas.

Many of the spallation products are short-lived and some decay through a chain of radioactive atoms. Several of the products are isotopes of mercury with decay chains consisting mainly of relatively short-lived progeny that are not usually encountered in dose assessments. Several of these decay chains have progeny with half-lives somewhat longer than their parent and comparable to the time required to travel from the SNS to potential receptors. As a result, the radiological characteristics of a plume of these spallation products can change significantly as it moves through the environment.

F.2 Source Terms for Normal and Accident Conditions

This section provides a summary discussion of source terms for normal and accident conditions at the SNS and tables listing source terms for individual radionuclides. A report providing the details of the bases for these source terms is included as Appendix A of this EIS.

F.2.1 Radionuclide Inventories

Radionuclide inventories used to derive source terms are based on a 1 MW beam power. Source terms for 4 MW operations assume that the specific activity (Ci/g, Ci/ml) of the materials released is four times the specific activity at 1 MW. Inventories for source terms for isotopes of mercury and iodine released from irradiated mercury assume that the SNS operates continuously at 1 MW beam power for 30 years with a single charge of mercury. Radionuclide inventories for source terms for other systems assume continuous operation at 1 MW for 1 year.

Both assumptions are conservative. When the particle beam is turned on, the activities of radionuclides begin to increase towards a "steady state" unique to each radionuclide and dependant on the beam power and intensity. Many nuclides reach a steady state after days, or even hours, of irradiation; however, some do not attain a steady state even after 30 years of continuous irradiation. The particle beam would be switched on and off many times over the 40-year life of the facility, and would be off much more than on; therefore, these inventories become increasingly conservative as the time necessary for a radionuclide to reach steady state increases. Inventories used to estimate source terms of specific radionuclides may be found in References 1 and 2 and in Appendix A of this EIS.

F.2.2 Normal Conditions

Source terms for annual emissions of normal operations from the Tunnel Confinement Exhaust Stack and the Target Building Exhaust Stack are shown in Table F-1. The base source terms were provided by the Department of Energy (DOE) (DeVore 1998b; DeVore 1998a) and have been adjusted when necessary for particle beam power. With the exception of mercury releases from the target cell (discussed below), DOE reduced radionuclide inventories by an availability factor of 0.559. This factor assumes that the beam is on 85 percent of the 240 days per year that the SNS is projected to be in use.

Assumptions on facility design are presented in the Conceptual Design Report (ORNL 1997a). For upgrade from 1 MW to 4 MW, a linear scaling of off-gases from the cooling system and the target are anticipated. Off-gases from the beam stops and exhausts from the various tunnels through the Tunnel Confinement Exhaust do not scale linearly, because of specifics of the proposed upgrade design.

F.2.2.1 Tunnel Confinement Exhaust

Radionuclides discharged from the Tunnel Confinement Exhaust Stack are gases and concrete dust particles activated as a result of beam interactions in the tunnels. Only a few have half-lives as long as a few minutes. It was estimated that, on average, 28.5 seconds would elapse between activation and discharge of the air (DeVore 1998a). The source term shown in Table F-1 reflects this decay.

F.2.2.2 Target Building Exhaust

Source terms for releases from the Target Building Exhaust include the affects of radioactive decay ingrowth, off-gas treatment, and HEPA filtration.

Table F-1
Projected Annual Emissions of Radionuclides from SNS Facilities During Normal Operations.

	Target Building Exhaust (Ci)						Tunnel Confinement Exhaust (Ci)	
	Cooling Systems[a]		Target Off-Gas[a]		Beam Stops[b]		Linac, Ring, and Beam Transfer Tunnels[b]	
Nuclides[c]	**1 MW**	**4 MW**	**1 MW**	**4 MW**	**1 MW**	**4 MW**	**1 MW**	**4 MW**
H-3	2.76E-00	1.11E+01	2.24E+01	8.96E+01	2.39E-00	4.46E-00	1.22E-07	1.22E-07
He-6	0	0	0	0	0	0	1.50E-08	2.36E-08
Li-8	0	0	0	0	0	0	1.31E-08	1.73E-08
Be-7	3.14E-03	1.26E-02	0	0	0	0	0	0
Be-10	2.62E-10	1.05E-09	0	0	0	0	0	0
C-10	0	0	0	0	0	0	2.55E+01	4.04E+01
C-11	0	0	0	0	0	0	4.06E+01	6.04E+01
C-14	1.33E-01	5.31E-01	0	0	1.37E-02	2.56E-02	1.08E-04	1.08E-04
N-13	0	0	0	0	0	0	3.18E+02	4.83E+02
N-16	0	0	0	0	0	0	7.92E-00	1.15E+01
O-14	0	0	0	0	0	0	8.99E+01	1.33E+02
O-15	0	0	0	0	0	0	3.41E+02	5.19E+02
F-18	5.85E-10	2.34E-09	0	0	0	0	0	0
F-20	0	0	0	0	0	0	2.97E-02	2.97E-02
Ne-23	0	0	0	0	0	0	1.90E-02	1.90E-02
Na-22	2.07E-08	8.29E-08	0	0	0	0	1.12E-02	1.12E-02
Na-24	0	0	0	0	0	0	2.46E-00	2.46E-00
Mg-27	0	0	0	0	0	0	1.05E-01	1.05E-01
Al-26	3.99E-13	1.60E-12	0	0	0	0	1.69E-06	1.69E-06
Al-28	0	0	0	0	0	0	8.61E-00	8.61E-00
Al-29	0	0	0	0	0	0	2.70E-02	2.70E-02
Si-31	0	0	0	0	0	0	7.34E-01	7.34E-01
Si-32	2.78E-10	1.11E-09	0	0	0	0	0	0
P-32	3.43E-08	1.37E-07	0	0	0	0	0	0
P-33	1.85E-09	7.40E-09	0	0	0	0	0	0
S-35	9.03E-09	3.61E-08	0	0	0	0	0	0
Cl-36	5.58E-12	2.23E-11	0	0	0	0	1.81E-06	1.81E-06
Cl-38	0	0	0	0	0	0	5.21E-04	5.21E-04
Ar-37	1.26E+02	5.02E+02	0	0	2.50E+02	4.67E+02	3.81E-01	3.81E-01
Ar-39	1.46E-01	5.83E-01	0	0	2.06E-01	3.85E-01	1.27E-02	1.27E-02
Ar-41	0	0	0	0	0	0	9.70E-04	9.70E-04
Ar-42	7.87E-02	3.15E-01	0	0	2.66E-02	4.97E-02	1.05E-06	1.05E-06
K-38	0	0	0	0	0	0	7.02E-04	7.02E-04
K-40	2.90E-15	1.16E-14	0	0	0	0	3.15E-07	3.15E-07

Table F-1
Projected Emissions of Radionuclides from SNS Facilities During Normal Operations. (Continued)

	Target Building Exhaust (Ci)						Tunnel Confinement Exhaust (Ci)	
	Cooling Systems[a]		Target Off-Gas[a]		Beam Stops[b]		Linac, Ring, and Beam Transfer Tunnels[b]	
Nuclides[c]	**1 MW**	**4 MW**	**1 MW**	**4 MW**	**1 MW**	**4 MW**	**1 MW**	**4 MW**
K-42	5.91E-13	2.37E-12	0	0	0	0	1.00E-00	1.00E-00
K-43	1.46E-12	5.85E-12	0	0	0	0	2.94E-04	2.94E-04
K-44	0	0	0	0	0	0	5.44E-04	5.44E-04
Ca-41	7.33E-11	2.93E-10	0	0	0	0	3.16E-03	3.16E-03
Ca-45	3.36E-08	1.35E-07	0	0	0	0	7.30E-01	7.30E-01
Ca-47	1.72E-10	6.90E-10	0	0	0	0	1.56E-03	1.56E-03
Ca-49	0	0	0	0	0	0	8.00E-02	8.00E-02
Sc-43	2.75E-22	1.10E-21	0	0	0	0	0	0
Sc-44	1.06E-21	4.23E-21	0	0	0	0	0	0
Sc-46	1.42E-07	5.70E-07	0	0	0	0	0	0
Sc-47	1.94E-08	7.77E-08	0	0	0	0	1.57E-03	1.57E-03
Sc-48	1.30E-09	5.19E-09	0	0	0	0	0	0
Sc-49	0	0	0	0	0	0	7.97E-02	7.97E-02
Ti-44	1.24E-08	4.97E-08	0	0	0	0	0	0
Ti-45	2.97E-26	1.19E-25	0	0	0	0	0	0
V-48	1.86E-06	7.45E-06	0	0	0	0	0	0
V-49	4.10E-06	1.64E-05	0	0	0	0	0	0
V-50	3.06E-22	1.22E-21	0	0	0	0	0	0
Cr-48	1.87E-10	7.49E-10	0	0	0	0	0	0
Cr-51	2.34E-04	9.35E-04	0	0	0	0	3.42E-04	3.42E-04
Mn-52	4.10E-06	1.64E-05	0	0	0	0	3.21E-05	3.21E-05
Mn-53	1.27E-10	5.07E-10	0	0	0	0	7.49E-09	7.49E-09
Mn-54	1.33E-05	5.30E-05	0	0	0	0	5.15E-03	5.15E-03
Mn-56	1.34E-28	5.35E-28	0	0	0	0	5.85E-03	5.85E-03
Fe-52	3.00E-14	1.20E-13	0	0	0	0	0	0
Fe-55	3.24E-04	1.29E-03	0	0	0	0	5.69E-01	5.69E-01
Fe-59	7.07E-06	2.83E-05	0	0	0	0	1.72E-02	1.72E-02
Fe-60	2.96E-13	1.18E-12	0	0	0	0	0	0
Co-55	4.87E-09	1.95E-08	0	0	0	0	0	0
Co-56	4.91E-05	1.96E-04	0	0	0	0	0	0
Co-57	1.15E-04	4.60E-04	0	0	0	0	0	0
Co-58	4.09E-05	1.64E-04	0	0	0	0	0	0
Co-60	5.11E-06	2.05E-05	0	0	0	0	0	0
Ni-56	1.03E-06	4.11E-06	0	0	0	0	0	0

Table F-1
Projected Emissions of Radionuclides from SNS Facilities During Normal Operations. (Continued)

	Target Building Exhaust (Ci)						Tunnel Confinement Exhaust (Ci)	
	Cooling Systems[a]		Target Off-Gas[a]		Beam Stops[b]		Linac, Ring, and Beam Transfer Tunnels[b]	
Ni-57	7.30E-07	2.92E-06	0	0	0	0	0	0
Ni-59	2.06E-06	8.23E-06	0	0	0	0	0	0
Ni-63	2.56E-04	1.02E-03	0	0	0	0	0	0
Ni-65	5.82E-26	2.33E-25	0	0	0	0	0	0
Cu-61	6.07E-25	2.43E-24	0	0	0	0	0	0
Cu-64	9.94E-14	3.98E-13	0	0	0	0	0	0
Sb-119	0	0	2.42E-02	9.67E-02	0	0	0	0
Te-119	0	0	1.67E-02	6.70E-02	0	0	0	0
Te-121	0	0	2.38E-02	9.53E-02	0	0	0	0
Te-123	0	0	1.61E-01	6.43E-01	0	0	0	0
I-121	0	0	4.96E-26	1.98E-25	0	0	0	0
I-122	0	0	5.22E-04	2.09E-03	0	0	0	0
I-123	0	0	4.43E-04	1.77E-03	0	0	0	0
I-124	0	0	5.69E-04	2.27E-03	0	0	0	0
I-125	0	0	3.91E-02	1.56E-01	0	0	0	0
I-129	0	0	3.58E-10	1.43E-09	0	0	0	0
I-130	0	0	1.76E-05	7.05E-05	0	0	0	0
Xe-122	0	0	1.04E-00	4.17E-00	0	0	0	0
Xe-123	0	0	1.72E-23	6.87E-23	0	0	0	0
Xe-125	0	0	1.18E-00	4.71E-00	0	0	0	0
Xe-127	0	0	8.05E+01	3.22E+02	0	0	0	0
Hg-192	0	0	1.19E-02	4.77E-02	0	0	0	0
Hg-193	0	0	4.84E-03	1.94E-02	0	0	0	0
Hg-194	0	0	2.25E-02	9.01E-02	0	0	0	0
Hg-195	0	0	1.21E-01	4.84E-01	0	0	0	0
Hg-197	0	0	3.60E-00	1.44E+01	0	0	0	0
Hg-203	0	0	3.29E-00	1.32E+01	0	0	0	0
Total	1.29E+02	5.15E+02	1.12E+02	4.50E+02	2.52E+02	4.72E+02	8.37E+02	1.26E+03

[a] DeVore 1998i.
[b] DeVore 1998h.
[c] Nuclides with activities of less than 1.0×10^{-30} Ci are not shown.

F.2.2.3 Cooling Water Systems

The source term for cooling water systems (DeVore 1998b) includes the contributions of D_2O and H_2O cooling water systems in the Target Building and H_2O cooling water systems in the beam stops. It includes two components: off-gas consisting of H-3 vapor and gaseous radionuclides, and mist from cooling water assumed to be at 90ºF. The mist was assumed to contain entrained activated metal corrosion products from the systems being cooled and to have the same radionuclide concentrations as the liquid low-level waste (see Section 4, Appendix A).

Mist eliminators in the system were assumed to have an efficiency of 70 percent. Emissions were assumed to occur over a 24-hour period, each time quarterly maintenance would be performed. Radionuclides emissions would be decayed for a total of 8 days before release (24 hours of emission evolution and 7 days hold-up in the decay tank). The total annual emissions are shown in Table F-1.

F.2.2.4 Beam Stop Emissions

Beam stop emissions were assumed to consist of activated air in the beam stop buildings and to be discharged via the gas decay tanks after 7 days total decay (DeVore 1998a). Emissions from cooling water systems in the beam stops are included in the previous source term.

F.2.2.5 Target Off-Gas Emissions

The source term for Target Off-Gas combines the tritium vapor, xenon gas, and mercury vapor in target off-gas with mercury vapor and mercuric iodide evaporating from mercury spilled in a target cell during target change-outs (DeVore 1998b). DOE assumed that iodine in the target would be chemically bound in non-volatile compounds of mercury.

Target off-gases would be collected and processed in the hot off-gas and off-gas decay systems. Air from the target cell would be vented through the cell ventilation system. The source term for mercury is based on its vapor pressure at –20º C, the temperature of the Mercury chiller/condenser, and off-gas system flow rate. The small quantity of mercury vapor that would not be condensed was assumed to decay for 7 days before release. The source term does not include the ingrowth of mercury progeny during this 7 days. Source terms for tritium and xenon were based on the quantities of these radionuclides generated in the first 10 seconds of irradiation. The quantities were corrected for decay of xenon and ingrowth of iodine over the 7 days required to fill a decay tank and the 7 additional days of decay before the tank would be discharged (DeVore 1998b). The tellurium and antimony progeny were assumed to be in equilibrium with their parents. It was assumed that HEPA filters and iodine absorbers would remove 99.95 percent of xenon progeny.

Mercury and mercuric iodide releases from the target cell were based on the vapor pressure of mercury at the temperatures and air flow rate in the cell. The mercury was assumed to be present as small droplets that accumulate each time the target mercury is replaced. The evaporation rate was based on the surface area of these droplets. It was assumed that there would be a 24-hour delay prior to each change-out to allow the system to cool completely and the short-lived radionuclides to decay. The availability factor was not applied to the target cell component.

Table F-2 Summary of SNS accident scenarios and source terms[a].

ID	Event	Hazard	Driving Force	Barriers[b]	Frequency[c]	Source Term[d]	Duration
A. Accidents Involving the SNS Target or Target Components							
1	Loss of Particle Beam focus or directional control (Appendix A, Section 3.1)	Radionuclides and Hg in target	Heating of target by proton beam	a) Automatic beam cutoff system	Anticipated	None	None
				b) Operator manual beam cutoff	Extremely Unlikely	Bounded by Event 3b	Bounded by Event 3b
2	Major loss of integrity of Hg Target Vessel or piping (Appendix A, Section 3.2)	Radionuclides and Hg in target	Hg pump	a) Automatic beam cutoff system, Mercury enclosure	Unlikely	Percent Inventory Mercury Iodine 0.002 0.002 0.14 0.14 0.142 0.142	 Interval 0 - 10 min 10 min - 3 days
				b) None	Extremely Unlikely	Percent Inventory Mercury Iodine 0.015 0.015 0.19 33 0.038 67 0.243 100.	 Interval 0 - 10 min 10 min - 10 days 10 - 30 days
3	Loss of Hg flow in Target (Appendix A, Section 3.3)	Radionuclides and Hg in target	Heating of target by proton beam	a) Automatic beam cutoff system	Anticipated	None	None
				b) Operator manual beam cutoff	Beyond Extremely Unlikely	Bounded by Event 16	Bounded by Event 16

Table F-2 Summary of SNS accident scenarios and source terms[a] - Continued.

ID	Event	Hazard	Driving Force	Barriers[b]	Frequency[c]	Source Term[d]	Duration
4	Loss of water flow in Hg Target Heat Exchanger (Appendix A, Section 3.4)	Radionuclides and Hg in target	Heating of target by proton beam	a) Automatic beam cutoff system	Anticipated	None	None
				b) Operator manual cutoff	Extremely Unlikely	Bounded by Event 3b	Bounded by Event 3b
5	Loss of water flow in Target Cooling Shroud (Appendix A, Section 3.5)	Radionuclides in target cooling water	Heating of target by proton beam	a) Automatic beam cutoff system	Anticipated	None	None
				b) Operator manual beam cutoff	Extremely Unlikely	Bounded by Event 8	Bounded by Event 8
6	Loss of water flow to Proton Beam Window (Appendix A, Section 3.6)	Radionuclides in cooling water	Heating of window by proton beam	a) Automatic beam cutoff system	Anticipated	None	None
				b) Operator manual beam cutoff	Extremely Unlikely	Bounded by Event 8	Bounded by Event 8
7	Loss of water flow to Target Component Cooling Loop (Appendix A, Section 7)	Radionuclides in cooling water	Heating of core vessel components by proton beam	a) Automatic beam cutoff system	Anticipated	None	None
				b) Operator manual cutoff	Unlikely	Bounded by Event 8	Bounded by Event 8

Table F-2 Summary of SNS accident scenarios and source terms[a] - Continued.

ID	Event	Hazard	Driving Force	Barriers[b]	Frequency[c]	Source Term[d]	Duration
				c) None	Extremely Unlikely	Bounded by Event 16	Bounded by Event 16
8	Loss of integrity in Target Component Cooling Loop (Appendix A, Section 3.8)	Radionuclides in cooling water	Heating of core vessel components by proton beam	a) Stack monitor	Anticipated	Bounded by annual release limits	Bounded by annual release limits
				b) Complete evaporation (utility vault)	Anticipated	Gases + Mist + 150 L of D_2O	5 min 30 min
				c) Complete evaporation (core vessel)	Anticipated	18 L of D_2O	30 days
				d) Complete evaporation	Anticipated	Gases + Mist + 150 L of H_2O	5 min 30 min
9	Loss of integrity in Cryogenic Moderator (Appendix A, Section 3.9)	Hydrogen gas	Hydrogen pressure in moderator system	None	Extremely Unlikely	No radionuclides.	Not specified
10	Loss of Core Vessel integrity (Appendix A, Section 3.10)	Activated air	Helium pressure in system	None	Unlikely	Not specified	Not specified
11	Loss of He flow to Core Vessel (Appendix A, Section 3.11)	Activated air	Helium pressure in system	None	Anticipated	Not specified	Not specified
12	Loss of Target Cell Ventilation (Appendix A, 3.12)	Mercury and radionuclides in Hg off-gas	Gaseous diffusion	None	Anticipated	Not specified	Not specified

Table F-2 Summary of SNS accident scenarios and source terms[a] - Continued.

ID	Event	Hazard	Driving Force	Barriers[b]	Frequency[c]	Source Term[d]	Duration
13	Loss of Offsite Power (Appendix A, Section 3.13)	Not specified	None	See Events 1 through 12	Not specified	Bounded by Events 1 through 12	Bounded by Events 1 through 12
14	Fire (Appendix A, Section 3.14)	See Events 1 through 12	Heating and/or Events 1 through 12	See Events 1 through 12	Not specified	Bounded by Events 1 through 13	Bounded by Events 1 through 13
15	Natural Phenomena (Appendix A, Section 3.15)	Mercury and radionuclides in target, radionuclides in cooling water, activated air	Tornadoes and earthquakes	None	Unlikely	Bounded by Events 1 through 14	Bounded by Events 1 through 14
16	Beyond Design Basis Hg Spill (Appendix A, Section 3.16)	Radionuclides and Hg in target	a) Heating by 1-MW proton beam plus decay heat	None	Beyond Extremely Unlikely	Percent Inventory Mercury Iodine 0.0066 14.0 0.80 20.0 0.30 60.0 1.11 100.	 Interval 0 - 10 min 1 - 7 days 7 - 30 days
		Radionuclides and Hg in target	b) Heating by 4-MW proton beam plus decay heat	None	Beyond Extremely Unlikely	Percent Inventory Mercury Iodine 0.183 14.0 0.800 20.0 0.300 60.0 1.28 100.	 Interval 0 - 10 min 1 - 7 days 7 - 30 days
			B. Accidents Involving SNS Waste Systems				
17	Hg Condenser Failure (Appendix A, Section 4.1.1)	Hg radionuclides in off-gas	Offgas blowers	None	Anticipated	13.7 g mercury	48 hours

Table F-2 Summary of SNS accident scenarios and source terms[a] - Continued.

ID	Event	Hazard	Driving Force	Barriers[b]	Frequency[c]	Source Term[d]	Duration
18	Hg Charcoal Absorber Failure [e] (Appendix A, Section 4.1.2)	Hg radionuclides in offgas	Offgas blowers	Stack monitor	Unlikely	14.8 g mercury	10 days
19	He Circulator Failure (Appendix A, Section 4.2.1)	Tritium in offgas	Offgas blowers	Circulator replacement	Anticipated	1 day tritium production	24 hours
20	Oxidation of Getter Bed (Appendix A, Section 4.2.2)	Tritium in offgas	Offgas blowers	Bed replacement	Unlikely	1 day tritium production	24 hours
21	Combustion of Getter Bed (Appendix A, Section 4.3.1)	Tritium absorbed on bed, depleted uranium in bed.	Combustion	Complete combustion	Extremely Unlikely	1 year tritium production, 200 g depleted uranium	1 hour
22	Failure of Cryogenic Charcoal Absorber [f] (Appendix A, Section 4.4.1)	Noble gases and iodine	Offgas blowers	System repair	Unlikely	1 day xenon production	24 hours
23	Valve sequence error in Tritium Removal System (Appendix A, Section 4.5.1)	Tritium accumulated in system	Offgas blowers	None	Unlikely	1 year tritium production	20 min
24	Valve sequence error in Offgas Decay System (Appendix A, Section 4.5.2)	Radionuclides accumulated in decay tank	Offgas blowers	None	Anticipated	7 days xenon accumulation (1 decay tank)	1 hour

Table F-2 Summary of SNS accident scenarios and source terms[a] - Continued.

ID	Event	Hazard	Driving Force	Barriers[b]	Frequency[c]	Source Term[d]	Duration
25	Spill during filling of tanker truck for LLLW[g] Storage Tanks (Appendix A, Section 4.5.3)	Radionuclides in tank	Evaporation and diffusion	Tank vault and HEPA filters	Anticipated	0.00005% of contents of LLLW tank	1 hour
26	Spray during filling of tanker truck for LLLW[g] (Appendix A, Section 4.5.6)	Radionuclides in tank	Pressure in transfer pipe	Operator cutoff and HEPA filters	Anticipated	1.9 mil of LLLW	20 min
27	Spill during filling of tanker truck for Process Waste Storage Tanks[g] (Appendix A, Section 4.5.5)	Radionuclides in tank.	Transfer pump	None	Anticipated	51,100 L Process Waste to surface water + 57 L to atmosphere	3.5 hours
28	Spray during filling of tanker truck for Process Waste[g] (Appendix A, Section 4.5.7)	Radionuclides in tank	Pressure in transfer pipe	Operator cutoff	Anticipated	28.4 L of Process Waste	20 min
29	Offgas Treatment pipe break (Appendix A, Section 4.6.1)	Radionuclides in target offgas	Cell ventilation blowers	Pipe repair	Unlikely	24 hours xenon production	24 hours
30	Offgas Compressor Failure (Appendix A, Section 4.6.2)	Radionuclides in target offgas.	Cell ventilation blowers	Compressor repair	Unlikely	1 hour xenon production	1 hour

Table F-2 Summary of SNS accident scenarios and source terms[a] - Continued.

ID	Event	Hazard	Driving Force	Barriers[b]	Frequency[c]	Source Term[d]	Duration
31	Offgas Decay Tank Failure (Appendix A, Section 4.6.3)	Radionuclides in target offgas	Cell ventilation blowers	None	Extremely Unlikely	7 days xenon accumulation	1 min
32	Offgas Charcoal Filter Failure (Appendix A, Section 4.6.4)	Iodine radionuclides in target offgas	Offgas blowers	None	Unlikely	7 days iodine production	24 hours
33	LLLW System piping failure (Appendix A, Section 4.6.5)	Radionuclides in waste.	Pumping	Linac tunnel and HEPA filters	Unlikely	0.00005% of contents of LLLW tank	1 hour
34	LLLW Storage Tank Failure (Appendix A, Section 4.6.6)	Radionuclides in tank	Gravity	Tank vault and HEPA filters	Extremely Unlikely	0.00005% of contents of LLLW tank	1 hour
35	LLLW pump failure (Appendix A, Section 4.6.7)	Radionuclides in waste	Gravity	Backup pumps and pump containment	Anticipated	None	None
36	Process Waste System piping failure (Appendix A, Section 4.6.8)	Radionuclides in waste	Pumping	None	Anticipated	10% of annual flow (no airborne release specified)	1 year
37	Process Waste Storage Tank Failure (Appendix A, Section 4.6.9)	Radionuclides in tank	Gravity	Dike/sump	Extremely Unlikely	57 L to atmosphere	1 hour
38	Process Waste System pump failure (Appendix A, Section 4.6.10)	Radionuclides in waste	Gravity	Backup pumps and pump containment?	Anticipated	None	None

Table F-2 Summary of SNS accident scenarios and source terms[a] - Continued.

ID	Event	Hazard	Driving Force	Barriers[b]	Frequency[c]	Source Term[d]	Duration
39	LLLW Transportation Accident [g] (Appendix A, Section 4.7.1)	Radionuclides in 800 gal LR-56 tanker truck	Collision/gravity	None	Extremely Unlikely 1.8×10^{-8}/trip 1.0×10^{-6}/year	800 gal LLLW (no airborne release specified)	24 hours
40	Process Waste Transportation Accident [g] (Appendix A, Section 4.7.2)	Radionuclides in 15,000 gal tanker truck	Collision/gravity	None	Unlikely 1.8×10^{-6} /trip 2.0×10^{-5}/year	15,000 gal process waste (no airborne release specified.	1 hour

[a] This table was compiled as a summary of information prepared by Lockheed Martin Energy Research (LMER) (refer to Sections 3.0 and 4.0 of Appendix A).

[b] The barriers listed are those that are assumed to prevent or terminate the release of radioactive or hazardous materials. Generally, one or more additional barriers such as HEPA filters or automatic alarms are present but have been ignored to increase the conservatism of the estimated source terms.

[c] Refer to Table 5.1.9-2 for the numerical ranges associated with accident frequencies categories.

[d] Source terms are expressed in units that are independent of power level. Except for Beyond Design Basis accidents (ID 16a, 16b), the radioactivity released in accidents at 4 MW is four times that released at 1 MW.

[e] Installation of sulfur-impregnated charcoal filters is being considered to serve as a “polishing filter” for the Mercury Condenser (refer to Event 17).

[f] Cryogenic charcoal absorbers are being considered as an alternative to the offgas compressor, decay storage tanks, and ambient temperature charcoal filters (see Events 24, 30, 31, and 32).

[g] Accidents involving tanker truckers are applicable for an SNS facility at ORNL where liquid wastes would be trucked to existing facilities for treatment but may not be applicable for a facility at LANL, ANL, or BNL. Frequencies may differ based on the size of tankers and distances traveled at the other sites.

F.2.3 Accident Conditions

A total of 40 accident scenarios are described in Appendix A and summarized in Table F-2. This is not an indication that the proposed SNS would be a particularly accident-prone facility, but is the result of the rigorous hazard analysis that DOE requires even for low-hazard facilities such as the proposed SNS. Since the proposed SNS is still in the conceptual design stage and dose estimates had not been made previously for these potential accidents, the full set of accident scenarios has been retained in this EIS. Secondary stages of some accidents are conservatively assumed to last from 7 to 30 days, while in reality, administrative and emergency response actions would more probably terminate the release in a shorter time period.

The bases for the source terms used for accident conditions are discussed in Sections 3.0 and 4.0 of Appendix A of this EIS. The source terms in Appendix A do not always explicitly show the activity of each radionuclide. This is done here in Tables F-3 through F-11 for accident scenarios that release radioactive materials to the atmosphere. Each table assigns an accident ID, identifies the section of Appendix A where the basis for the source term is discussed, lists the nature and frequency of occurrence of the accident event, lists the duration and total activities of each radionuclide released in each stage of the accident, and lists the total duration and activities for each accident.

All source terms discussed in this section would be released from the Target Building Exhaust Stack except for that for the LLLW pipe break in the linac tunnel (Tunnel Confinement Exhaust Stack) and all process waste source terms (ground-level releases assumed near the Target Building).

F.2.3.1 Mercury Spills

Table F-3 lists source terms for spills of irradiated mercury that could occur within the limits established by the design basis for the target system. The activities shown are for a beam power of 1 MW and would be four times greater at a beam power of 4 MW. Table F-4 lists source terms for beyond-design-basis spills at power levels of 1 MW and 4 MW. In addition to the 4:1 ratio in activities, the 4 MW source term assumes boiling of the mercury during the first stage of the accident (refer to Exhibit F of Appendix A). Both sets of source terms are bounding source terms for reasonably foreseeable mercury spills that could occur within or beyond the design basis.

The radionuclide activities shown in these tables reflect adjustment of the source terms from Appendix A to account for radioactive decay. Decay to the mid-point of the cumulative accident duration at the end of each phase was used to approximate the average release rate for each phase. Since the model for these source terms assumes that only mercury and mercuric iodide are volatile, their progeny are not included in the source terms; however, they were taken into account in the transport and dose calculations. (Sections F.4 and F.5).

F.2.3.2 Cooling Water System Leaks

Bounding source terms for accident involving leaks in the D_2O and H_2O cooling systems are listed in Table F-5. Leaks in the Utility Vault are assumed to be rapid (i.e., pipe breaks) so that dissolved gases would be released suddenly. The leak in the Core Vessel is assumed to be a slow leak so that dissolved gases are released at essentially the same rate as under normal conditions and can, therefore, be ignored. The activities shown correspond to the beginning of the release. Decay to the appropriate mid-points was performed during transport calculations.

F.2.3.3 Off-Gas Decay System Failures

Bounding source terms for accidents involving failures of the Off-Gas Decay System are listed in Table F-6. Cryogenic Charcoal failure is included here since the primary function of this device is to condense and hold relatively short-lived radionuclides until they decay. It is a alternative to the decay tanks. Source terms involving the Decay Tank (ID 24, 31) were assumed to occur immediately after the tank is filled. These source terms account for decay of xenon and ingrowth of iodine as the tank is filled and assume that tellurium and antimony progeny are in equilibrium with their iodine parents. All activities correspond to the beginning of the release. Decay during release is accounted for in the transport calculations.

Table F-3.
Source Terms for Design Basis Target Mercury Spill Scenarios.

ID [a]				2a				2b
Section [b]				3.2				3.2
Event				Spill Contained in Hg Enclosure				Spill Not Contained in Hg Enclosure
Probability [c]				Unlikely				Extremely Unlikely
Duration [d] **(sec)**	600	690,600	0	691,200	600	863,400	1,728,000	2,592,000
Nuclide	**Ci**	**Ci**	**Ci**	**Total Ci**	**Ci**	**Ci**	**Ci**	**Total Ci**
I-119	1.16E-04	0	0	1.16E-04	8.72E-04	0	0	8.72E-04
I-120	1.95E-04	5.54E-24	0	1.95E-04	1.46E-03	5.81E-27	0	1.46E-03
I-121	3.97E-04	6.65E-16	0	3.97E-04	2.98E-03	6.13E-17	0	2.98E-03
I-122	2.59E-04	0	0	2.59E-04	1.94E-03	0	0	1.94E-03
I-123	7.40E-04	3.45E-04	0	1.09E-03	5.55E-03	2.32E-02	1.13E-09	2.88E-02
I-124	3.39E-04	1.33E-02	0	1.37E-02	2.54E-03	2.72E+00	7.34E-01	3.46E+00
I-125	1.49E-03	9.92E-02	0	1.01E-01	1.11E-02	2.31E+01	3.98E+01	6.30E+01
I-126	6.75E-05	3.83E-03	0	3.89E-03	5.06E-04	8.55E-01	8.28E-01	1.68E+00
I-128	6.01E-05	0	0	6.01E-05	4.51E-04	0	0	4.51E-04
I-129	1.77E-10	1.24E-08	0	1.26E-08	1.33E-09	2.92E-06	5.93E-06	8.85E-06
I-130	3.37E-05	1.09E-05	0	4.46E-05	2.53E-04	6.68E-04	8.89E-12	9.21E-04
Hg-180	1.38E-29	0	0	1.38E-29	1.03E-28	0	0	1.03E-28
Hg-181	5.86E-25	0	0	5.86E-25	4.39E-24	0	0	4.39E-24
Hg-182	7.58E-11	0	0	7.58E-11	5.68E-10	0	0	5.68E-10
Hg-183	1.26E-11	0	0	1.26E-11	9.42E-11	0	0	9.42E-11
Hg-184	8.27E-06	0	0	8.27E-06	6.20E-05	0	0	6.20E-05
Hg-185	1.14E-04	0	0	1.14E-04	8.56E-04	0	0	8.56E-04
Hg-186	1.25E-03	0	0	1.25E-03	9.40E-03	0	0	9.40E-03
Hg-187	6.25E-03	0	0	6.25E-03	4.69E-02	0	0	4.69E-02
Hg-188	1.96E-02	0	0	1.96E-02	1.47E-01	0	0	1.47E-01
Hg-189	5.02E-02	0	0	5.02E-02	3.77E-01	0	0	3.77E-01
Hg-190	9.24E-02	0	0	9.24E-02	6.93E-01	0	0	6.93E-01
Hg-191	1.27E-01	0	0	1.27E-01	9.49E-01	0	0	9.49E-01
Hg-192	1.77E-01	1.42E-05	0	1.77E-01	1.33E+00	6.29E-07	1.96E-28	1.33E+00
Hg-193	2.06E-01	3.72E-07	0	2.06E-01	1.54E+00	6.37E-09	0	1.54E+00
Hg-194	2.26E-02	1.58E+00	0	1.61E+00	1.70E-01	2.15E+00	4.30E-01	2.75E+00
Hg-195	3.46E-01	2.91E-02	0	3.75E-01	2.59E+00	7.34E-03	8.67E-14	2.60E+00
Hg-197	2.32E+00	5.76E+01	0	5.99E+01	1.74E+01	6.03E+01	3.20E-01	7.81E+01
Hg-203	1.65E+00	1.09E+02	0	1.11E+02	1.24E+01	1.46E+02	2.37E+01	1.82E+02
Hg-205	4.10E-02	0	0	4.10E-02	3.07E-01	0	0	3.07E-01
Total	5.06E+00	1.68E+02	0	1.73E+02	3.80E+01	2.35E+02	6.58E+01	3.39E+02

[a] Accident identification number from Table 5.1.9-3.
[b] Section number of Appendix A of this EIS.
[c] See Table 5.1.9-2 for numerical ranges corresponding to description.
[d] Time over which activity is released for an accident scenario. Release occurs in more than one phase for some scenarios.

Table F-4
Source Terms for Beyond Design Basis Target Mercury Spill Scenarios.

ID[a]				16a				16b
Section[b]				3.16				3.16
Event	Loss of Hg Flow/Delayed Beam Cutoff (1 MW)				Loss of Hg Flow/Delayed Beam Cutoff (4 MW)			
Probability[c]	Reasonably Foreseeable				Reasonably Foreseeable			
Duration[d] **(sec)**	600	604,200	1,987,200	2,592,000	600	604,200	1,987,200	2,592,000
Nuclide	**Ci**	**Ci**	**Ci**	**Total Ci**	**Ci**	**Ci**	**Ci**	**Total Ci**
I-119	8.14E-01	0	0	8.14E-01	3.26E+00	0	0	3.26E+00
I-120	1.36E+00	3.75E-19	0	1.36E+00	5.45E+00	1.50E-18	0	5.45E+00
I-121	2.78E+00	4.80E-12	0	2.78E+00	1.11E+01	1.92E-11	0	1.11E+01
I-122	1.81E+00	0	0	1.81E+00	7.25E+00	0	0	7.25E+00
I-123	5.18E+00	9.23E-02	1.69E-10	5.27E+00	2.07E+01	3.69E-01	6.77E-10	2.11E+01
I-124	2.37E+00	2.05E+00	5.83E-01	5.00E+00	9.48E+00	8.18E+00	2.33E+00	2.00E+01
I-125	1.04E+01	1.43E+01	3.85E+01	6.32E+01	4.16E+01	5.70E+01	1.54E+02	2.53E+02
I-126	4.73E-01	5.61E-01	7.54E-01	1.79E+00	1.89E+00	2.24E+00	3.01E+00	7.15E+00
I-128	4.21E-01	0	0	4.21E-01	1.68E+00	0	0	1.68E+00
I-129	1.24E-06	1.77E-06	5.84E-06	8.85E-06	4.95E-06	7.08E-06	2.34E-05	3.54E-05
I-130	2.36E-01	3.05E-03	1.16E-12	2.39E-01	9.45E-01	1.22E-02	4.65E-12	9.57E-01
Hg-180	4.54E-29	0	0	4.54E-29	5.04E-27	0	0	5.04E-27
Hg-181	1.93E-24	0	0	1.93E-24	2.14E-22	0	0	2.14E-22
Hg-182	2.50E-10	0	0	2.50E-10	2.77E-08	0	0	2.77E-08
Hg-183	4.15E-11	0	0	4.15E-11	4.60E-09	0	0	4.60E-09
Hg-184	2.73E-05	0	0	2.73E-05	3.03E-03	0	0	3.03E-03
Hg-185	3.76E-04	0	0	3.76E-04	4.17E-02	0	0	4.17E-02
Hg-186	4.14E-03	0	0	4.14E-03	4.59E-01	0	0	4.59E-01
Hg-187	2.06E-02	0	0	2.06E-02	2.29E+00	0	0	2.29E+00
Hg-188	6.46E-02	0	0	6.46E-02	7.17E+00	0	0	7.17E+00
Hg-189	1.66E-01	0	0	1.66E-01	1.84E+01	0	0	1.84E+01
Hg-190	3.05E-01	0	0	3.05E-01	3.38E+01	0	0	3.38E+01
Hg-191	4.18E-01	7.63E-29	0	4.18E-01	4.63E+01	3.05E-28	0	4.63E+01
Hg-192	5.84E-01	4.49E-04	9.14E-30	5.85E-01	6.48E+01	1.80E-03	3.65E-29	6.48E+01
Hg-193	6.79E-01	1.89E-05	0	6.79E-01	7.53E+01	7.56E-05	0	7.53E+01
Hg-194	7.47E-02	9.05E+00	3.39E+00	1.25E+01	8.28E+00	3.62E+01	1.36E+01	5.80E+01
Hg-195	1.14E+00	3.85E-01	5.49E-14	1.53E+00	1.26E+02	1.54E+00	2.20E-13	1.28E+02
Hg-197	7.66E+00	3.75E+02	1.71E+00	3.84E+02	8.49E+02	1.50E+03	6.85E+00	2.36E+03
Hg-203	5.45E+00	6.27E+02	1.83E+02	8.15E+02	6.04E+02	2.51E+03	7.31E+02	3.84E+03
Hg-205	1.35E-01	0	0	1.35E-01	1.50E+01	0	0	1.50E+01
Total	4.26E+01	1.03E+03	2.28E+02	1.30E+03	1.96E+03	4.11E+03	9.10E+02	6.98E+03

Table F-5
Source Terms for Target Cooling Water Systems Failures.

ID[a]			8b			8c			8d
Section[b]			3.8			3.8			3.8
Event			Heavy Water Leak in Utility Vault			Heavy Water Leak in Core Vessel			Light Water Leak in Utility Vault
Probability[c]			Anticipated			Anticipated			Anticipated
Duration[d] **(sec)**	300	1,800	2,100	2,592,000	0	2,592,000	300	1,800	2,100
Nuclide	**Ci**	**Ci**	**Total Ci**	**Ci**	**Ci**	**Total Ci**	**Ci**	**Ci**	**Total Ci**
H-3	1.88E+01	1.88E+02	2.06E+02	2.25E+01	0	2.25E+01	1.89E+00	1.88E+01	2.06E+01
Be-7	1.62E-03	0	1.62E-03	0	0	0	1.62E-03	0	1.62E-03
C-14	1.39E-05	0	1.39E-05	0	0	0	1.39E-05	0	1.39E-05
N-13	1.09E+02	0	1.09E+02	0	0	0	1.09E+02	0	1.09E+02
O-14	6.40E+00	0	6.40E+00	0	0	0	6.40E+00	0	6.40E+00
O-15	1.43E+02	0	1.43E+02	0	0	0	1.43E+02	0	1.43E+02
V-49	1.38E-05	0	1.38E-05	0	0	0	1.38E-05	0	1.38E-05
Mn-54	4.39E-05	0	4.39E-05	0	0	0	4.39E-05	0	4.39E-05
Fe-55	1.39E-03	0	1.39E-03	0	0	0	1.39E-03	0	1.39E-03
Fe-59	2.44E-06	0	2.44E-06	0	0	0	2.44E-06	0	2.44E-06
Co-56	5.24E-05	0	5.24E-05	0	0	0	5.24E-05	0	5.24E-05
Co-57	3.59E-04	0	3.59E-04	0	0	0	3.59E-04	0	3.59E-04
Co-58	3.68E-05	0	3.68E-05	0	0	0	3.68E-05	0	3.68E-05
Co-60	2.33E-05	0	2.33E-05	0	0	0	2.33E-05	0	2.33E-05
Ni-63	1.24E-03	0	1.24E-03	0	0	0	1.24E-03	0	1.24E-03
Total	2.77E+02	1.88E+02	4.65E+02	2.25E+01	0	2.25E+01	2.60E+02	1.88E+01	2.79E+02

Table F-6
Source Terms for Off-Gas Decay System Failure Scenarios.

ID[a]	22	24	29	30	31
Section[b]	4.4.1	4.5.2	4.6.1	4.6.2	4.6.3
Event	Cryogenic Charcoal Failure	Decay Tank Valve Sequence Error	Off-Gas Pipe Break	Off-Gas Compressor Failure	Decay Tank Failure
Probability[c]	Unlikely	Anticipated	Unlikely	Unlikely	Extremely Unlikely
Duration[d] **(sec)**	86,400	3,600	86,400	3,600	60
Nuclide	**Ci**	**Ci**	**Ci**	**Ci**	**Ci**
H-3	1.10E-01	7.69E-01	1.10E-01	4.58E-03	7.69E-01
C-10	4.38E-03	3.07E-02	4.38E-03	1.83E-04	3.07E-02
C-11	3.23E-01	2.26E+00	3.23E-01	1.35E-02	2.26E+00
C-14	1.62E-04	1.14E-03	1.62E-04	6.77E-06	1.14E-03
N-13	1.36E+00	9.51E+00	1.36E+00	5.66E-02	9.51E+00
N-16	1.23E-02	8.63E-02	1.23E-02	5.14E-04	8.63E-02
O-14	3.29E-01	2.30E+00	3.29E-01	1.37E-02	2.30E+00
O-15	6.14E+00	4.30E+01	6.14E+00	2.56E-01	4.30E+01
Ar-37	1.80E-01	1.26E+00	1.80E-01	7.51E-03	1.26E+00
Ar-39	1.78E-04	1.25E-03	1.78E-04	7.42E-06	1.25E-03
Ar-41	4.63E-03	3.24E-02	4.63E-03	1.93E-04	3.24E-02
Ar-42	9.59E-05	6.71E-04	9.59E-05	4.00E-06	6.71E-04
Sb-119	4.49E-07	3.23E+00	4.49E-07	1.87E-08	3.23E+00
Te-119	2.61E-03	3.23E+00	2.61E-03	1.09E-04	3.23E+00
Te-121	9.59E-07	1.69E+00	9.59E-07	4.00E-08	1.69E+00
Te-123m	4.58E-07	1.14E+01	4.58E-07	1.91E-08	1.14E+01
I-119	2.92E+01	3.23E+00	2.92E+01	1.22E+00	3.23E+00
I-120	6.11E-01	1.78E+00	6.11E-01	2.55E-02	1.78E+00
I-121	3.81E-01	1.69E+00	3.81E-01	1.59E-02	1.69E+00
I-122	2.64E+00	1.18E+01	2.64E+00	1.10E-01	1.18E+01
I-123	1.37E-01	1.14E+01	1.37E-01	5.71E-03	1.14E+01
I-125	4.74E-04	2.47E+01	4.74E-04	1.97E-05	2.47E+01
Xe-119	4.50E+02	3.23E+00	4.50E+02	1.87E+01	3.23E+00
Xe-120	4.26E+01	1.78E+00	4.26E+01	1.77E+00	1.78E+00
Xe-121	4.15E+01	1.69E+00	4.15E+01	1.73E+00	1.69E+00
Xe-122	9.62E+00	1.18E+01	9.62E+00	4.01E-01	1.18E+01
Xe-123	9.28E+01	1.14E+01	9.28E+01	3.87E+00	1.14E+01
Xe-125	3.52E+01	3.67E+01	3.52E+01	1.47E+00	3.67E+01
Xe-127	4.77E-01	3.17E+00	4.77E-01	1.99E-02	3.17E+00
Total	7.13E+02	2.03E+02	7.13E+02	2.97E+01	2.03E+02

Table F-7
Source Terms for Mercury Removal System Failure Scenarios.

ID [a]	17	18
Section [b]	4.1.1	4.1.2
Event	Hg Condensor Failure	Hg Charcoal Absorber Failure
Probability [c]	Anticipated	Unlikely
Duration [d] **(sec)**	172,800	864,000
Nuclide	**Ci**	**Ci**
Hg-184	1.20E-04	1.30E-04
Hg-185	1.91E-04	2.06E-04
Hg-186	5.25E-04	5.68E-04
Hg-187	1.11E-03	1.20E-03
Hg-188	2.47E-03	2.67E-03
Hg-189	4.29E-03	4.63E-03
Hg-190	5.43E-03	5.87E-03
Hg-191	6.84E-03	7.40E-03
Hg-192	9.01E-03	9.74E-03
Hg-193	9.77E-03	1.06E-02
Hg-194	5.71E-04	6.17E-04
Hg-195	1.77E-02	1.91E-02
Hg-197	1.18E-01	1.28E-01
Hg-203	8.46E-02	9.15E-02
Hg-205	3.64E-03	3.94E-03
Total	2.65E-01	2.86E-01

Table F-8
Source Terms for Tritium Removal System Failure Scenarios.

ID[a]	19	20	21	23
Section[b]	4.2.1	4.2.2	4.3.1	4.5.1
Event	He Circulator Failure	Oxidation of Tritium Getter Bed	Combustion of Tritium Getter Bed	Valve Sequence Error
Probability[c]	Anticipated	Unlikely	Extremely Unlikely	Unlikely
Duration[d] **(sec)**	86,400	86,400	3,600	1,200
Nuclide	**Ci**	**Ci**	**Ci**	**Ci**
H-3	4.58E-01	4.58E-01	4.00E+03	4.00E+03
U-234	0	0	1.25E-05	0
U-235	0	0	8.48E-07	0
U-236	0	0	3.88E-07	0
U-238	0	0	8.10E-05	0
Total	4.58E-01	4.58E-01	4.00E+03	4.00E+03

Table F-9
Source Term for Iodine Removal System Failure Scenario.

ID [a]	32
Section [b]	4.6.4
Event	Off-Gas Charcoal Filter Failure
Probability [c]	Unlikely
Duration [d] **(sec)**	86,400
Nuclide	**Ci**
I-119	2.92E+01
I-120	6.11E-01
I-121	3.81E-01
I-122	2.64E+00
I-123	1.37E-01
I-125	4.74E-04
Total	3.29E+01

Table F-10
Source Terms for Liquid Low-Level Waste System Failure Scenarios.

ID [a]	25	26	33	34
Section [b]	4.5.3	4.5.6	4.6.5	4.6.6
Event	Spill Filling Tanker Truck	Spray Filling Tanker Truck	Pipe Break in Linac Tunnel	Storage Tank Failure
Probability [c]	Anticipated	Anticipated	Unlikely	Extremely Unlikely
Duration [d] **(sec)**	3,600	1,200	3,600	3,600
Nuclide	**Ci**	**Ci**	**Ci**	**Ci**
H-3	4.96E-03	2.48E-02	4.96E-03	4.96E-03
Be-7	2.03E-05	1.01E-06	2.03E-05	2.03E-05
C-14	1.74E-07	8.71E-09	1.74E-07	1.74E-07
V-49	1.73E-07	8.65E-09	1.73E-07	1.73E-07
Mn-54	5.48E-07	2.74E-08	5.48E-07	5.48E-07
Fe-55	1.74E-05	8.68E-07	1.74E-05	1.74E-05
Fe-59	3.04E-08	1.52E-09	3.04E-08	3.04E-08
Co-56	6.55E-07	3.27E-08	6.55E-07	6.55E-07
Co-57	4.49E-06	2.24E-07	4.49E-06	4.49E-06
Co-58	4.60E-07	2.30E-08	4.60E-07	4.60E-07
Co-60	2.91E-07	1.46E-08	2.91E-07	2.91E-07
Ni-63	1.55E-05	7.73E-07	1.55E-05	1.55E-05
Total	5.02E-03	2.48E-02	5.02E-03	5.02E-03

Table F-11
Source Terms for Liquid Process Waste System Failure Scenarios.

ID[a]	27	28	37
Section[b]	4.5.5	4.5.7	4.6.9
Event	Storage Tank Failure	Spray Filling Tanker Truck	Spill Filling Tanker Truck
Probability[c]	Extremely Unlikely	Anticipated	Anticipated
Duration[d] **(sec)**	12,600	1,200	3,600
Nuclide	**Ci**	**Ci**	**Ci**
H-3	7.31E-05	3.66E-05	7.31E-05
Be-7	5.53E-05	2.77E-05	5.53E-05
C-14	5.01E-08	2.51E-08	5.01E-08
V-48	6.92E-09	3.46E-09	6.92E-09
V-49	4.52E-08	2.26E-08	4.52E-08
Cr-51	1.53E-08	7.65E-09	1.53E-08
Mn-52	1.40E-07	7.33E-09	1.40E-07
Mn-54	2.17E-12	1.09E-12	2.17E-12
Fe-55	5.94E-08	2.97E-08	5.94E-08
Fe-59	5.09E-06	2.54E-06	5.09E-06
Co-56	1.57E-07	7.87E-08	1.57E-07
Co-57	9.01E-07	4.50E-07	9.01E-07
Co-58	1.91E-06	9.53E-07	1.91E-06
Co-60	7.69E-07	3.84E-07	7.69E-07
Ni-59	0	2.56E-07	0
Ni-63	3.58E-08	1.79E-08	3.58E-08
Total	1.38E-04	6.90E-05	1.38E-04

F.2.3.4 Off-Gas Treatment System Failures

Tables F-7 through F-9 list bounding source terms for accidents involving failures of systems designed to remove mercury, tritium, and iodine from target off-gas. The Mercury Charcoal Absorber (Table F-7) is not currently part of the design but may be added if conditions warrant.

F.2.3.5 Liquid Low-Level Waste (LLLW) System Failures

Bounding source terms for failures of the LLLW System that result in releases to the atmosphere are listed in Table F-10. These source terms are the only ones that assume filtration by HEPA filters. All activities correspond to the beginning of the release. Decay during release is accounted for in the transport calculations.

F.2.3.6 Process Waste System Failures

Bounding source terms for failures of the Process Waste System that result in releases to the atmosphere are listed in Table F-11. All activities correspond to the beginning of the release. Decay during release is accounted for in the transport calculations.

F.2.3.7 Source Terms Not Considered

All of the source terms discussed in the preceding subsection are released directly to the atmosphere and were used in evaluating health impacts in this EIS. Appendix A includes four accident scenarios that involve direct releases to soil. One of these accidents also includes a release to surface water as well as a release to air. The release to air was included. This subsection provides the basis for excluding these additional source terms from consideration.

Section 4.5.5 of Appendix A discusses an “anticipated” spill of the contents of a Process Waste Storage Tank. The airborne source term for this accident is included in Table F-11. The scenario also assumes that 13,500 gal of process waste overflows the curb around the tank, enters the retention basin, and enters the receiving stream. The discharge points of the retention basins at the other SNS alternative sites are not specified. Other accident scenarios assume that only members of the public beyond the ORR boundary and boundaries of the other sites would be exposed. In addition, this EIS only considers exposures that are an immediate result of accidents (Section F.3). Accordingly, only the airborne source term applicable to all sites has been included in the health impacts assessment.

Section 4.6.8 of Appendix A discusses an “anticipated” break of an underground process waste pipe that releases 10 percent of the annual volume of process waste underground. It is assumed that the leak is discovered after one year. The scenario does not postulate that the liquid released pools on the surface of the ground or enters the groundwater system or discuss the depth of soil over the release. Since there is no surface pooling, the radioactivity released could reach humans only via groundwater transport. Any radionuclides would move in the direction of groundwater flow. Tritium would migrate at the velocity of groundwater flow and C-14 at a somewhat slower rate. Migration of other radionuclides in the waste would move much more slowly and could require many years to reach a location where human exposure could occur. Most of these radionuclides would decay to negligible concentrations before such migration could occur.

Section 4.7.1 of Appendix A discusses a transportation accident involving the release of LLLW from the LR-56 tanker truck and Section 4.7.2 discusses a similar accident involving process waste. Both accidents assume a total loss of tanker contents but do not postulate airborne release. The LR-56 is essentially a DOT Type B transport package with a capacity of 800 gallons but is not certified as such in the United States. No radioactive material has ever been released in a transportation accident involving a certified Type B package. The process waste tanker has a capacity of 15,000 gallons and no special resistance to severe transportation accidents. Based on the annual number of trips, the LLLW accidents would be "extremely unlikely" and the process waste accident would be "unlikely." In the absence of an airborne source term, it is unlikely that humans would be accidentally exposed before the spill was immobilized and assessed, and any appropriate remedial actions taken.

F.3 Selection of Exposure Pathways

This section identifies the potential pathways for exposure of human to radioactive materials that would or could be released from the SNS and discusses the rationale for selecting these pathways. This information is also applicable to assessment of the toxic effects of exposures to mercury.

Summary

This EIS evaluates health impacts of normal operations and accidents based on two exposure pathways for workers and the public:

- Inhalation of radionuclides released to air.
- Immersion in air containing radionuclides released to air.
- Ingestion of foods contaminated by radionuclides released to air.

For accidents, the ingestion pathway would be a delayed impact and impacts could, therefore, be controlled by impoundment of foodstuffs and by remedial actions.

Discussion

Radioactive materials released during normal and accident conditions may be released to air, soil, surface water, and/or groundwater. Each of these media have a number of primary and secondary exposure pathways that may be important. Which exposure pathways are important depends on the radiological characteristics of the radionuclides and the quantities of each released and on how the radionuclides would be diluted or concentrated as they are transferred from one medium or pathway to another.

All radioactive and toxic materials released to the environment during normal SNS operations are released to the atmosphere. The majority of the releases are continuous throughout the year. Under these conditions, the primary potential exposure pathways and groups exposed are:

- Inhalation of radionuclides released to air (workers, public),

- Immersion in air containing radionuclides released to air (workers, public), and

- Ingestion of foods contaminated by radionuclides released to air (public).

The ingestion pathway could include a number of sub-pathways. Radionuclides deposited on the surfaces of leafy plants could be absorbed by the plants and radionuclides deposited on the ground surface could be taken up by the roots of plants. Once in the plants, the radionuclides could be ingested by humans eating the plants, and/or eating animals that had eaten the plants, or by humans eating products such as milk or eggs from animals that had eaten the plants.

Potential secondary exposure pathways for releases to air involve radionuclides deposited on the ground surface. The pathways and the groups exposed are:

- Exposure to direct radiation from radionuclides deposited on the ground surface (workers, public),

- Inhalation of resuspended contaminated soil (workers, public), and

- Immersion in air containing resuspended contaminated soil (workers, public).

Doses from the secondary exposure pathways are usually much lower and often insignificant compared to doses from the primary pathways. The relative importance of the primary pathways to each other depends more directly on the specific radionuclides released.

These same potential exposure pathways exist for accidental releases; however, because accidental releases occur infrequently and over relatively short periods of time, the relative importance of pathways based on deposition of radionuclides on the ground surface is diminished. Radionuclides deposited on plants or the ground surface are removed by weathering and would not be replenished. In case of large accidental releases, the site emergency response plan may involve actions to prevent ingestion of contaminated foods and to remove contamination from the environment. Based on these considerations, accident impacts were evaluated in this EIS based on exposures of workers and the public via inhalation and immersion only.

An extensive EPA assessment of mercury exposure (EPA 1997) investigated atmospheric deposition of mercury. It found that the combined wet and dry deposition of elemental mercury vapor on the ground was very low and that approximately 5 to 10 percent of mercuric mercury (oxidized mercury) would be deposited within 100 km of the release point. It also found that elemental mercury was rarely absorbed by the leafy surfaces or root of plants. SNS source terms for normal emissions assume that all mercury would be released as elemental mercury vapor. Some accident scenarios do assume that iodine would be released as mercuric iodide, an oxidized mercury, but the amount of mercury released in this form would be many orders of magnitude less than the quantity of elemental mercury.

F.4 Environmental Transport

The assessment of health impacts in this EIS is based on evaluation of the consequences of elevated and ground-level releases of radioactive and toxic materials from the SNS. The materials released would be transported through the environment by atmospheric dispersion. During dispersion, additional factors could affect the concentrations of contaminants in the air. These plume depletion mechanisms include dry deposition (“fallout”), wet deposition (“rainout” and “washout”), and radioactive decay.

A number of computer codes are available to calculate dispersion, deposition, and radioactive decay of radionuclides released to the atmosphere and many of these codes also calculate transport of deposited radionuclides through the food chain. CAP88-PC is a widely-used code that performs such calculations for continuous releases such as SNS emissions in routine operations. GENII and MACCS2 can perform these calculations for both continuous and short-duration releases that would occur during accidents. None of these codes contain decay chain data, biotic transfer factors, or dose conversion factors for some of the mercury, xenon, and iodine radionuclides and associated progeny produced in the mercury target, and it would not be practical to make the necessary modifications to the codes and their data files.

F.4.1 Undepleted Atmospheric Dispersion Factors

For normal conditions, a set of Microsoft Excel97 spreadsheet and Visual Basic macros were developed to implement a slightly modified version of the methodology used in CAP88-PC. This methodology is described in the code user guide (EPA 402-B-92-001). The documentation for AIRDOS-EPA, a mainframe predecessor of CAP88-PC, contains additional detail and a source code listing (EPA 520/1/79-009).

The CAP88-PC methodology implemented in this analysis uses a Gaussian plume model to calculate sector-averaged deleted ground-level concentrations in air and the ground deposition rates of radionuclides. The depletion mechanisms considered are radioactive decay and ingrowth, precipitation scavenging, and dry deposition. In-growth of progeny of radionuclides deposited on the ground and on plant surfaces are also considered. Concentrations in vegetation, beef, and milk consumed by humans are calculated using soil-to-plant, animal feed-to-milk, and animal feed-to-beef transfer factors. Intake of radionuclides by humans is calculated based on agricultural production data for the appropriate state and consumption rates of leafy vegetables, produce, milk, and beef.

The following modifications were made to the CAP88-PC methodology:

- Plume rise was conservatively assumed to be zero.

- Dose and risk calculations and data were replaced by updated dose conversion factors discussed in Section F.5.2 and risk factors recommended by the ICRP.

- The CAP88-PC consideration of ingrowth of a small number of decay chains and the use of pre-calculated ingrowth factors in decay and buildup calculations were replaced with specific calculation of ingrowth of all decay chains.

- The time allowed for deposition and buildup of radionuclides was changed from 100 years to 40 years to match the operating life of the SNS.

- The maximally exposed individual was assumed to be a hypothetical individual located at the site boundary and to obtain all of his or her required dietary intake at this location. The CAP88-PC method of adjusting the relative amounts of food grown in a given segment, grown in the entire assessment area, and imported from outside the region that is ingested by the population in that segment was retained for population dose calculations.

- When calculating population doses, CAP88-PC determines the maximally exposed individual based only on results for segments that are specified in the population distribution as containing people. For this analysis, a hypothetical individual was placed in the sector where contamination would have the maximum impact on agricultural production in the region of the assessment [i.e., within 50 mi (80 km) of the site].

F.4.2 Depletion by Radioactive Decay – Normal Operations

Site-specific joint frequency distributions in STAR format were used to calculate the wind speed frequencies and averages and the stability class frequencies required for the CAP88-PC methodology. Site-specific precipitation data and atmospheric lid heights were used in dispersion and deposition calculations. Dry deposition rates for particulates (0.035 m/sec), iodine (0.0018 m/sec), and gases (0 m/sec) listed in the CAP88-PC user's guide were used; however, a deposition velocity of 0.0006 m/sec (ref 3) was used for mercury.

CAP88-PC biotic transfer factors were supplemented with data from ORNL-5786 (Baes 1984) and from http://risk.lsd.ornl.gov/cgi-bin/tox/TOX_9801. The CAP88-PC methodology uses transfer factors for vegetation consumed by humans based on the wet weight of the vegetation. ORNL-5786 contains factors based on dry weight but provides a conversion factor for adapting the data for use with CAP88-PC. Agricultural production data for Tennessee, New Mexico, Illinois, and New York were used in site-specific evaluations.

The analysis used CAP88-PC default values for fractions of vegetables, beef, and milk consumed by populations. Fractions assumed to be grown locally, in the assessment region, and imported were the CAP88-PC defaults for rural areas for ORNL and LANL and for urban areas for ANL and BNL. CAP88-PC consumption rates were also used. Site-specific populations distributions were used for the off-site public and for uninvolved workers.

F.4.3 Accident Conditions

Atmospheric dispersion calculations for short-term releases in accidents were performed using PAVAN, a computer code used by the U.S. Nuclear Regulatory Commission to evaluate ground-level concentrations of radioactive materials released in accidents at nuclear power plants (PNL 1982). PAVAN uses joint frequency distributions of wind speed and direction by stability class to calculate ground-level normalized atmospheric dispersion factors (ADFs or χ/Qs) for short-term elevated and ground-level releases. The code does not consider plume rise, radioactive decay, or any other depletion process. The short-term ADFs are normalized ground-level concentrations at the plume centerline in each 22.5 degrees sector surrounding the site.

PAVAN uses several methods to deal with the fact that meteorological conditions during a given short-term release will vary from release to release. For this EIS, direction-specific χ/Qs that would be exceeded no more than 0.5 percent of the total time were selected for short-term releases. PAVAN calculates sets of these χ/Qs for release durations of 0-2 hours, 0-8 hours, 8-24 hours, 1-4 days, and 4-30 days.

The wind speed, wind direction, and stability class data were for the most recent available one-year alternating period from the meteorological monitoring station nearest to the preferred SNS location at each site. ORNL provided 1996 data measured at heights of 10 m and 60 m at the Y-12 Plant western meteorological tower. LANL provided 1996 data measured at height of 10 m at the TA-53 tower. ANL provided 1997 data measured at a height of 60 m. BNL provided 1997 data measured at a height of 10 m. If 60 m data was available, it was used for elevated releases. Otherwise, 10 m data was used. PAVAN adjusts all wind speed data from the height of measurement to the height of release (10 m for ground-level releases).

For elevated releases, χ/Qs were calculated for 22.5 degree sectors centered on the principal compass directions. Distances spaced at increasing intervals from 100 m to 2 km were used for workers. Distances from each stack to the site boundary were used for the maximally exposed member of the public. Distances corresponding to those provided in offsite population distributions within 80 km of the site as provided by each site were used for the offsite populations calculation. Ground-level releases were assumed to occur near the Target Building Exhaust Stack. For uninvolved worker populations, χ/Qs were estimated by superimposing the 100-2000 m grid for individual workers on site maps. Worker populations in occupied structures were provided by ORNL and estimated for the other sites by querying electronic copies of site phone books.

The calculations for normal operations used 8-24 hour χ/Qs for releases from the cooling systems and annual average χ/Qs for other normal releases. The releases were modeled as elevated releases from the appropriate SNS stack. The heights of these stacks would be 80 feet above grade. No adjustments were made for terrain height. The calculations for accident conditions used the durations and source terms shown in Tables F-3 through F-11 and selected χ/Qs appropriate to each phase.

F.4.4 Depletion by Radioactive Decay – Accidental Releases

The spreadsheet macros also accounted for changes in concentrations of radionuclide in the plume due to radioactive decay and ingrowth during transport and, in the case of accidents, during release. This involved calculations for as many as 245 radionuclides. Many of these radionuclides have half-lives comparable to their travel times from the SNS to a distance of 80 km. Thus, the concentration and dose were very sensitive to distance. Elevated releases travel some distance, usually a few hundred meters, before the plume reaches the ground. As a result, χ/Qs initially increase and then begin to decrease with distance. For the radionuclides that would be emitted by the SNS, the total activity in the plume decreases with distance but activities of a number of progeny increase to some steady state or peak and then decline. This behavior can cause shifts in the relative importance of exposure pathways as the plume traverses the region of interest.

Since average wind speeds are not uniform in all directions, the spreadsheet macros used average wind speeds specific to each direction at a given site to calculate “in-flight” decay. These average wind speeds were calculated from joint frequency distributions of height-adjusted wind speeds and direction by stability class calculated by PAVAN from the original joint frequency distributions for each site.

The depleted uranium component of the source term for a fire in the tritium getter bed was not decayed. The half-lives of the uranium isotopes and their progeny is such that the progeny that have high dose conversion factors relative to the parent uranium require several thousand years to in-grow to levels that would affect dose.

F.5 Dose Calculations

This section discusses the calculation of dose to workers and the public from exposure to SNS emissions by inhalation and immersion, the selection of dose conversion factors from available data, and the basis for estimating ingestion dose to the public for inhalation dose.

F.5.1 Inhalation and Immersion

The total dose (rem) to an individual at a given distance and direction from the source of an airborne release due to radionuclide concentrations in the environment is given by:

$$Dose = \sum_{i=1}^{n} Q_i \cdot \frac{\chi}{Q} \cdot E \cdot \left(BR \cdot DCF_{inh_i} + DCF_{imm_i}\right)$$

where:

Qi	=	Depleted source term (Ci/sec) for the i-th radionuclide
χ/Q	=	Atmospheric dispersion factor (sec/m3) for the given distance, direction, and release duration
E	=	Exposure period (sec)
BR	=	Breathing rate (m3/sec)
DCF_{inh}	=	Inhalation dose conversion factor for the i-th radionuclide (rem/Ci)
DCF_{imm}	=	Immersion dose conversion factor for the i-th radionuclide (rem/sec per Ci/m3)
CR	=	Consumption rate (grams/day)
DCF_{ingi}	=	Ingestion dose conversion factor for the i-th radionuclide (rem/sec per g/d)

For exposures to continuous releases, exposure periods are 8,760 hr/yr for the public and 2,000 hr/yr for workers. For short-term releases, the exposure period for the public is equal to the release duration. For workers, it is the number of hours worked during the release based on 8-hours shifts starting at the beginning of the release. Dose conversion factors for inhalation and immersion are listed in Table F-12 and discussed in Section F.5.2.

F.5.2 Selection of Dose Conversion Factors

Most dose assessments use dose conversion factors published by the U.S. Environmental Protection Agency in *Federal Guidance Report No. 11* (Eckerman et al 1988) for internal exposures and *Federal Guidance Report No. 12* (Eckerman and Ryman 1993) for external exposures. The factors are applicable to exposures received by workers and the public and are reflected in current dose limits enforced by EPA, DOE, and NRC. These reports were the primary source of the dose conversion factor used to prepare this EIS; however, they do not include data for all of the mercury and iodine radionuclides or their progeny that are projected to be present in SNS emissions.

Table F-12
Dose Conversion Factors Used to Estimate SNS Impacts under Normal and Accident Conditions

Nuclide	Half Life	Inhalation Rem/Ci	Immersion Rem/sec per Ci/m3	Ground Plane Rem/sec per Ci/m2	Ingestion Rem/Ci
H-3	12.6 y	6.40E+01	0	0	6.40E+01
He-6	0.81 s	#N/A	#N/A	#N/A	#N/A
Li-8	0.84 s	#N/A	#N/A	#N/A	#N/A
Be-7	53.3 d	3.21E+02	8.73E-03	1.81E-04	1.28E+02
Be-8	0.00 s	#N/A	#N/A	#N/A	#N/A
Be-10	1.55E+06 y	3.54E+05	4.14E-05	1.52E-06	4.66E+03
B-12	0.20 s	#N/A	#N/A	#N/A	#N/A
B-13	0.02 s	#N/A	#N/A	#N/A	#N/A
C-10	19.3 s	#N/A	#N/A	#N/A	#N/A
C-11	20.4 m	1.22E+01	1.81E-01	3.74E-03	1.22E+01
C-14	5,870 y	2.09E+03	8.29E-07	5.96E-08	2.09E+03
N-12	0.01	#N/A	#N/A	#N/A	#N/A
N-13	9.97 m	#N/A	1.81E-01	3.74E-03	#N/A
N-16	7.13 s	#N/A	#N/A	#N/A	#N/A
N-17	4.17 s	#N/A	#N/A	#N/A	#N/A
O-14	1.18 m	#N/A	#N/A	#N/A	#N/A
O-15	2.04 m	#N/A	1.82E-01	3.74E-03	#N/A
O-19	26.9 s	#N/A	#N/A	#N/A	#N/A
F-18	1.83 h	8.36E+01	1.81E-01	3.74E-03	1.22E+02
F-20	11.0 s	#N/A	#N/A	#N/A	#N/A
Ne-23	37.2 s	#N/A	#N/A	#N/A	#N/A
Na-22	2.67 y	7.66E+03	4.00E-01	7.77E-03	1.15E+04

Nuclide	Half Life	Inhalation Rem/Ci	Immersion Rem/sec per Ci/m3	Ground Plane Rem/sec per Ci/m2	Ingestion Rem/Ci
Na-24	15.0 h	1.21E+03	8.07E-01	1.34E-02	1.42E+03
Mg-27	9.46 m	#N/A	#N/A	#N/A	#N/A
Al-26	7.59E+05 y	7.96E+04	5.03E-01	9.21E-03	1.46E+04
Al-28	2.24 m	#N/A	3.43E-01	5.99E-03	#N/A
Al-29	6.56 m	#N/A	#N/A	#N/A	#N/A
Si-31	2.62 H	2.23E+02	4.33E-04	1.11E-05	5.40E+02
Si-32	176 y	1.01E+06	1.94E-06	1.15E-07	2.18E+03
P-32	14.3 d	1.55E+04	3.66E-04	1.08E-05	8.77E+03
P-33	25.3 d	2.32E+03	3.05E-06	1.65E-07	9.18E+02
S-35	87.5 d	2.48E+03	8.99E-07	6.22E-08	7.33E+02
Cl-36	3.09E+05 y	2.19E+04	8.25E-05	2.49E-06	3.03E+03
Cl-38	37.2 m	1.34E+02	2.91E-01	4.96E-03	2.35E+02
Ar-37	35.0 d	#N/A	0	0	#N/A
Ar-39	276 y	#N/A	3.37E-05	1.25E-06	#N/A
Ar-41	1.82 h	#N/A	2.41E-01	4.44E-03	#N/A
Ar-42	33.7 y	#N/A	#N/A	#N/A	#N/A
Ar-43	5.37 m	#N/A	#N/A	#N/A	#N/A
K-38	7.64 m	#N/A	6.07E-01	1.08E-02	#N/A
K-40	1.31E+09 y	1.24E+04	2.98E-02	5.40E-04	1.86E+04
K-42	12.4 h	1.36E+03	5.40E-02	9.84E-04	1.13E+03
K-43	22.3 h	6.92E+02	1.73E-01	3.53E-03	7.70E+02
K-44	22.1 m	8.29E+01	4.40E-01	7.55E-03	1.73E+02

Table F-12
Dose Conversion Factors Used to Estimate SNS Impacts under Normal and Accident Conditions - Continued.

Nuclide	Half Life	Inhalation Rem/Ci	Immersion Rem/sec per Ci/m3	Ground Plane Rem/sec per Ci/m2	Ingestion Rem/Ci
Ca-41	1.06E+05 y	1.35E+03	0	0	1.27E+03
Ca-45	163 d	6.62E+03	3.19E-06	1.71E-07	3.16E+03
Ca-47	4.54 d	6.55E+03	1.98E-01	3.70E-03	6.51E+03
Ca-49	8.72 m	#N/A	6.40E-01	9.73E-03	#N/A
Sc-43	3.89 h	2.59E+02	1.95E-01	4.00E-03	7.62E+02
Sc-44	3.93 h	4.92E+02	3.89E-01	7.66E-03	1.43E+03
Sc-46	83.8 d	2.96E+04	3.69E-01	7.14E-03	6.40E+03
Sc-47	3.35 d	1.84E+03	1.90E-02	3.85E-04	2.23E+03
Sc-48	1.82 d	4.11E+03	6.22E-01	1.18E-02	7.25E+03
Sc-49	57.2 m	1.02E+02	7.14E-04	1.82E-05	2.52E+02
Ti-44	64.6 y	1.02E+06	2.05E-02	4.88E-04	2.31E+04
Ti-45	3.08 h	2.15E+02	1.55E-01	3.19E-03	5.99E+02
Ti-51	5.76 m	#N/A	#N/A	#N/A	#N/A
V-47	32.6 m	7.03E+01	1.77E-01	3.65E-03	1.75E+02
V-48	16.0 d	1.02E+04	5.37E-01	1.03E-02	8.58E+03
V-49	330 d	3.45E+02	0	0	6.14E+01
V-50	1.44E+17 y	#N/A	#N/A	#N/A	#N/A
V-52	3.74 m	#N/A	#N/A	#N/A	#N/A
Cr-48	21.6 h	8.77E+02	7.62E-02	1.57E-03	9.14E+02
Cr-49	42.3 m	7.25E+01	1.86E-01	3.85E-03	1.84E+02
Cr-51	27.7 d	3.34E+02	5.59E-03	1.14E-04	1.47E+02
Cr-55	3.50 m	#N/A	#N/A	#N/A	#N/A
Cr-56	5.94 m	#N/A	#N/A	#N/A	#N/A
Mn-51	46.2 m	1.15E+02	1.78E-01	3.67E-03	2.78E+02
Mn-52	5.59 d	5.70E+03	6.36E-01	1.22E-02	7.59E+03
Mn-53	3.83E+06 y	5.00E+02	0	0	1.08E+02
Mn-54	312 d	6.70E+03	1.51E-01	3.00E-03	2.77E+03
Mn-56	2.58 h	3.77E+02	3.19E-01	5.85E-03	9.77E+02
Mn-57	1.42 m	#N/A	#N/A	#N/A	#N/A
Fe-52	8.28 h	2.19E+03	1.31E-01	2.69E-03	5.59E+03
Fe-53	8.51 m	#N/A	#N/A	#N/A	#N/A
Fe-55	2.80 y	2.69E+03	0	0	6.07E+02
Fe-59	44.5 d	1.48E+04	2.21E-01	4.14E-03	6.70E+03
Fe-60	1.54E+06 y	7.47E+05	7.22E-07	5.48E-08	1.52E+05
Fe-61	5.98 m	#N/A	#N/A	#N/A	#N/A
Co-55	17.5 h	2.09E+03	3.62E-01	7.14E-03	4.37E+03
Co-56	77.3 d	3.96E+04	6.77E-01	1.22E-02	1.26E+04
Co-57	272 d	9.07E+03	2.08E-02	4.26E-04	1.18E+03
Co-58	70.9 d	1.09E+04	1.76E-01	3.52E-03	3.58E+03
Co-60	5.41 y	2.19E+05	4.66E-01	8.70E-03	2.69E+04
Co-61	1.65 h	1.06E+02	1.46E-02	3.34E-04	2.63E+02
Co-62	1.50 m	#N/A	#N/A	#N/A	#N/A

Table F-12
Dose Conversion Factors Used to Estimate SNS Impacts under Normal and Accident Conditions - Continued.

Nuclide	Half Life	Inhalation Rem/Ci	Immersion Rem/sec per Ci/m3	Ground Plane Rem/sec per Ci/m2	Ingestion Rem/Ci
Co-63	27.4 s	#N/A	#N/A	#N/A	#N/A
Ni-56	6.08 d	4.03E+03	3.11E-01	6.14E-03	3.89E+03
Ni-57	1.48 d	1.89E+03	3.59E-01	6.66E-03	3.77E+03
Ni-59	77,900 y	1.32E+03	0	0	2.10E+02
Ni-63	103 y	3.10E+03	0	0	5.77E+02
Ni-65	2.52 h	2.42E+02	1.03E-01	1.91E-03	6.22E+02
Cu-60	23.7 m	6.92E+01	7.33E-01	1.34E-02	1.93E+02
Cu-61	3.33 h	1.87E+02	1.48E-01	3.02E-03	4.37E+02
Cu-62	9.74 m	#N/A	1.80E-01	3.70E-03	#N/A
Cu-64	12.7 h	2.77E+02	3.37E-02	6.92E-04	4.66E+02
Sb-119	1.59 d	1.25E+02	7.96E-04	8.03E-05	2.75E+02
Te-119	16.0 h	3.76E+02	1.36E-01	2.76E-03	6.46E+02
Te-119m	4.70 d	#N/A	#N/A	#N/A	#N/A
Te-121	16.8 d	1.91E+03	9.99E-02	2.11E-03	1.68E+03
Te-123	1.03E+13 y	1.05E+04	7.96E-04	7.22E-05	4.18E+03
Te-123m	120 d	1.06E+04	2.41E-02	5.29E-04	5.66E+03
I-119	19.1 m	5.18E+01	1.57E-01	3.23E-03	1.48E+02
I-120	1.35 h	3.69E+02	5.11E-01	9.47E-03	1.27E+03
I-121	2.12 h	1.02E+02	7.18E-02	1.51E-03	3.08E+02
I-122	3.63 m	1.27E+01	1.69E-01	3.48E-03	4.78E+01
I-123	13.3 h	2.78E+02	2.69E-02	6.14E-04	8.05E+02
I-124	4.81 d	1.64E+04	1.99E-01	3.89E-03	4.81E+04

Nuclide	Half Life	Inhalation Rem/Ci	Immersion Rem/sec Per Ci/m3	Ground Plane Rem/sec per Ci/m2	Ingestion Rem/Ci
I-125	59.0 d	1.93E+04	1.93E-03	1.58E-04	5.69E+04
I-126	13.1 d	3.65E+04	7.96E-02	1.65E-03	1.07E+05
I-128	25.0 m	4.85E+01	1.54E-02	3.24E-04	1.70E+02
I-29	1.61E+07 y	1.33E+05	1.41E-03	9.55E-05	3.91E+05
I-130	12.4 h	2.50E+03	3.85E-01	7.77E-03	7.27E+03
Xe-119	5.80 m	#N/A	#N/A	#N/A	#N/A
Xe-120	40.0 m	#N/A	7.18E-02	1.57E-03	#N/A
Xe-121	40.1 m	#N/A	3.38E-01	6.25E-03	#N/A
Xe-122	20.1 h	#N/A	9.10E-03	2.53E-04	#N/A
Xe-123	2.08 h	#N/A	1.12E-01	2.25E-03	#N/A
Xe-125	16.9 h	#N/A	4.40E-02	9.81E-04	#N/A
Xe-127	36.4 d	#N/A	4.63E-02	1.01E-03	#N/A
Yb-169	32.1 d	8.07E+03	4.77E-02	1.12E-03	3.00E+03
Yb-169m	46.0 s	#N/A	#N/A	#N/A	#N/A
Lu-168	5.50 m	#N/A	#N/A	#N/A	#N/A
Lu-169	1.42 d	1.35E+03	1.88E-01	3.65E-03	2.03E+03
Lu-169m	2.67 m	#N/A	#N/A	#N/A	#N/A
Lu-170	2.01 d	2.58E+03	4.74E-01	8.29E-03	4.55E+03
Lu-172	6.70 d	5.00E+03	3.42E-01	6.70E-03	5.66E+03
Lu-172m	3.70 m	#N/A	#N/A	#N/A	#N/A
Lu-173	1.40 y	2.25E+04	1.89E-02	4.74E-04	1.09E+03

Table F-12
Dose Conversion Factors Used to Estimate SNS Impacts under Normal and Accident Conditions - Continued.

Nuclide	Half Life	Inhalation Rem/Ci	Immersion Rem/sec Per Ci/m3	Ground Plane Rem/sec per Ci/m2	Ingestion Rem/Ci
Hf-168	26.0 m	#N/A	#N/A	#N/A	#N/A
Hf-169	3.20 m	#N/A	#N/A	#N/A	#N/A
Hf-170	16.0 h	1.20E+03	9.32E-02	1.99E-03	2.12E+03
Hf-172	1.92 y	3.18E+05	1.50E-02	4.18E-04	4.48E+03
Hf-173	23.6 h	4.77E+02	6.85E-02	1.47E-03	1.00E+03
Hf-175	70.0 d	5.59E+03	6.25E-02	1.34E-03	1.82E+03
Ta-168	2.07 m	#N/A	#N/A	#N/A	#N/A
Ta-169	4.90 m	#N/A	#N/A	#N/A	#N/A
Ta-170	6.77 m	#N/A	#N/A	#N/A	#N/A
Ta-172	36.8 m	5.66E+01	2.81E-01	5.48E-03	1.59E+02
Ta-173	3.14 h	3.20E+02	1.02E-01	2.10E-03	7.84E+02
Ta-174	1.05 h	6.73E+01	1.10E-01	2.25E-03	1.96E+02
Ta-175	10.5 h	3.81E+02	1.68E-01	3.25E-03	9.07E+02
Ta-176	8.08 h	6.90E+02	4.14E-01	7.51E-03	1.12E+03
Ta-177	2.36 d	3.07E+02	9.36E-03	2.43E-04	4.51E+02
Ta-178	9.32 m	8.29E+01	#N/A	#N/A	2.93E+02
Ta-179	1.87 y	6.51E+03	4.03E-03	1.17E-04	2.73E+02
W-168	51.0 s	#N/A	#N/A	#N/A	#N/A
W-169	1.33 m	#N/A	#N/A	#N/A	#N/A
W-170	2.42 m	#N/A	#N/A	#N/A	#N/A
W-172	6.60 m	#N/A	#N/A	#N/A	#N/A
W-173	7.60 m	#N/A	#N/A	#N/A	#N/A
W-174	31.0 m	#N/A	#N/A	#N/A	#N/A
W-175	35.0 m	#N/A	#N/A	#N/A	#N/A
W-176	2.50 h	2.39E+02	2.60E-02	6.33E-04	3.60E+02
W-177	2.25 h	6.51E+01	1.58E-01	3.23E-03	2.48E+02
W-178	21.5 d	2.71E+02	1.71E-03	4.81E-05	1.02E+03
W-179	37.0 m	3.50E+00	6.77E-03	2.17E-04	1.01E+01
W-179m	6.40 m	#N/A	#N/A	#N/A	#N/A
W-181	122 d	5.00E+02	5.18E-03	1.46E-04	2.13E+02
Re-172	15.0 s	#N/A	#N/A	#N/A	#N/A
Re-172m	55.0 s	#N/A	#N/A	#N/A	#N/A
Re-173	1.98 m	#N/A	#N/A	#N/A	#N/A
Re-174	2.40 m	#N/A	#N/A	#N/A	#N/A
Re-175	5.88 m	#N/A	#N/A	#N/A	#N/A
Re-176	5.30 m	3.88E+01	1.91E-01	3.89E-03	8.40E+01
Re-177	14.0 m	2.39E+01	1.10E-01	2.18E-03	5.40E+01
Re-178	13.2 m	2.25E+01	2.25E-01	4.18E-03	5.77E+01
Re-179	19.5 m	#N/A	#N/A	#N/A	#N/A
Re-180	2.37 m	7.58E+00	2.10E-01	4.18E-03	7.39E+00
Re-181	19.8 m	9.16E+02	1.40E-01	2.88E-03	1.50E+03
Re-182m	12.7 h	#N/A	#N/A	#N/A	#N/A
Re-183	70.0 d	#N/A	#N/A	#N/A	#N/A
Os-172	19.2 s	#N/A	#N/A	#N/A	#N/A

Table F-12

Dose Conversion Factors Used to Estimate SNS Impacts under Normal and Accident Conditions - Continued.

Nuclide	Half Life	Inhalation Rem/Ci	Immersion Rem/sec per Ci/m3	Ground Plane Rem/sec per Ci/m2	Ingestion Rem/Ci
Os-173	16.0 s	#N/A	#N/A	#N/A	#N/A
Os-174	44.0 s	#N/A	#N/A	#N/A	#N/A
Os-175	1.40 m	#N/A	#N/A	#N/A	#N/A
Os-176	3.60 m	#N/A	#N/A	#N/A	#N/A
Os-177	2.80 m	#N/A	#N/A	#N/A	#N/A
Os-178	5.00 m	#N/A	#N/A	#N/A	#N/A
Os-179	6.50 m	#N/A	#N/A	#N/A	#N/A
Os-180	20.8 m	4.54E+01	5.77E-05	9.18E-06	5.44E+01
Os-181	1.75 h	6.71E+00	6.40E-02	1.32E-03	7.19E+00
Os-182	22.1 h	1.38E+03	7.44E-02	1.57E-03	2.44E+03
Os-183	13.0 h	9.72E+02	1.08E-01	2.28E-03	2.66E+03
Os-183m	9.89 h	#N/A	#N/A	#N/A	#N/A
Os-185	93.6 d	4.27E+03	1.22E-01	2.49E-03	1.77E+03
Os-186	2.05E+15 y	#N/A	#N/A	#N/A	#N/A
Os-189m	5.81 h	2.99E+01	3.92E-07	1.16E-07	6.70E+01
Ir-176	8.00 s	#N/A	#N/A	#N/A	#N/A
Ir-177	30.0 s	#N/A	#N/A	#N/A	#N/A
Ir-178	12.0 s	#N/A	#N/A	#N/A	#N/A
Ir-179	1.32 m	#N/A	#N/A	#N/A	#N/A
Ir-180	1.50 m	7.66E+00	1.58E-01	3.17E-03	6.83E+00
Ir-181	4.90 m	6.66E+01	7.59E-01	1.41E-02	5.67E+01
Ir-182	15.0 m	4.85E+01	2.41E-01	4.85E-03	1.28E+02
Ir-183	58.0 m	1.54E+02	2.11E-01	4.00E-03	3.03E+02
Ir-184	3.08 h	2.30E+02	3.47E-01	6.73E-03	6.96E+02
Ir-185	14.4 h	6.56E+02	1.09E-01	2.04E-03	9.72E+02
Ir-186	16.6 h	1.20E+03	2.96E-01	5.70E-03	1.97E+03
Ir-186m	1.90 h	#N/A	#N/A	#N/A	#N/A
Ir-187	10.5 h	2.53E+02	5.66E-02	1.18E-03	3.99E+02
Ir-188	1.72 d	1.66E+03	3.96E-01	7.03E-03	2.75E+03
Ir-189	13.2 d	1.69E+03	1.15E-02	2.76E-04	8.12E+02
Pt-176	6.33 s	#N/A	#N/A	#N/A	#N/A
Pt-177	11.0 s	#N/A	#N/A	#N/A	#N/A
Pt-178	21.1 s	#N/A	#N/A	#N/A	#N/A
Pt-179	21.2 s	#N/A	#N/A	#N/A	#N/A
Pt-180	52.0 s	6.27E+00	0	0	3.92E+00
Pt-181	51.0 s	2.56E+01	1.17E+00	2.26E-02	2.04E+01
Pt-182	2.20 m	#N/A	#N/A	#N/A	#N/A
Pt-183	6.50 m	#N/A	#N/A	#N/A	#N/A
Pt-183m	43.0 s	#N/A	#N/A	#N/A	#N/A
Pt-184	17.3 m	5.05E+01	1.17E-01	2.48E-03	4.45E+01
Pt-185	1.18 h	2.47E+02	5.03E-01	1.02E-02	2.38E+02
Pt-186	2.00 h	1.95E+02	1.14E-01	2.36E-03	3.27E+02
Pt-187	2.35 h	2.18E+02	9.77E-02	2.02E-03	2.62E+02
Pt-188	10.2 d	6.48E+03	3.35E-02	7.33E-04	3.00E+03
Pt-189	10.9 h	5.76E+02	8.29E-02	1.73E-03	6.86E+02

Table F-12
Dose Conversion Factors Used to Estimate SNS Impacts under Normal and Accident Conditions - Continued.

Nuclide	Half Life	Inhalation Rem/Ci	Immersion Rem/sec per Ci/m3	Ground Plane Rem/sec per Ci/m2	Ingestion Rem/Ci
Pt-190	6.66E+11 y	#N/A	#N/A	#N/A	#N/A
Pt-191	2.80 d	6.14E+02	4.96E-02	1.10E-03	1.46E+03
Pt-193	51.4 y	2.27E+02	1.47E-06	4.40E-07	1.19E+02
Au-180	8.10 s	#N/A	#N/A	#N/A	#N/A
Au-181	11.4 s	#N/A	#N/A	#N/A	#N/A
Au-182	15.6 s	#N/A	#N/A	#N/A	#N/A
Au-183	42.0 s	#N/A	#N/A	#N/A	#N/A
Au-184	53.0 s	2.66E+00	0	0	2.24E+00
Au-185	4.25 m	7.03E+01	1.88E-01	3.81E-03	8.22E+01
Au-186	10.7 m	8.04E+01	3.67E-01	7.25E-03	1.77E+02
Au-187	8.40 m	5.68E+01	1.88E-01	3.52E-03	3.22E+02
Au-187m	2.30 s	#N/A	#N/A	#N/A	#N/A
Au-188	8.83 m	#N/A	#N/A	#N/A	#N/A
Au-189	28.7 m	1.47E+02	6.66E-01	1.33E-02	1.93E+02
Au-190	42.8 m	7.20E+01	4.37E-01	7.66E-03	1.23E+02
Au-191	3.17 h	1.46E+02	1.00E-01	2.09E-03	1.87E+02
Au-191m	0.92 s	#N/A	#N/A	#N/A	#N/A
Au-192	4.94 h	3.27E+02	3.59E-01	6.44E-03	6.22E+02
Au-193	17.6 h	2.89E+02	2.53E-02	5.66E-04	5.77E+02
Au-194	1.59 d	1.02E+03	1.96E-01	3.70E-03	1.88E+03
Au-195	186 d	1.30E+04	1.19E-02	2.90E-04	1.06E+03
Au-195m	30.5 s	#N/A	3.47E-02	7.14E-04	#N/A
Hg-180	3.00 s	#N/A	#N/A	#N/A	#N/A
Hg-181	3.60 s	#N/A	#N/A	#N/A	#N/A
Hg-182	10.8 s	#N/A	#N/A	#N/A	#N/A
Hg-183	9.40 s	#N/A	#N/A	#N/A	#N/A
Hg-184	30.6 s	1.17E+02	1.03E-01	2.13E-03	3.06E+00
Hg-185	49.1 s	1.02E+03	0	0	1.66E+01
Hg-186	1.38 m	4.84E+01	6.99E-02	1.48E-03	2.56E+01
Hg-187	2.40 m	1.56E+03	7.73E-01	1.48E-02	1.05E+02
Hg-188	3.25 m	3.10E+01	3.54E-02	7.81E-04	3.68E+00
Hg-189	7.60 m	#N/A	#N/A	#N/A	#N/A
Hg-190	20.5 m	1.95E+02	3.05E-02	6.55E-04	6.39E+01
Hg-191	50.8 m	7.31E+02	2.62E-01	5.14E-03	1.61E+02
Hg-192	4.86 h	3.71E+03	4.66E-02	9.99E-04	8.28E+02
Hg-193	3.81 h	4.20E+03	3.22E-02	7.10E-04	3.09E+02
Hg-194	455 y	1.49E+05	2.56E-06	7.59E-07	5.13E+03
Hg-195	9.89 h	5.26E+03	3.40E-02	7.18E-04	3.63E+02
Hg-197	2.67 d	1.61E+04	9.84E-03	2.38E-04	8.67E+02
Hg-203	46.6 d	2.59E+04	4.18E-02	8.58E-04	1.99E+03
Hg-205	5.20 m	4.64E+01	9.21E-04	1.88E-05	3.09E+01
U-234	2.57E+05 y	1.32E+08	2.82E-05	2.77E-06	2.83E+05
U-235	7.40E+08 y	1.23E+08	2.66E-02	5.48E-04	2.66E+05
U-236	2.46E+07 y	1.25E+08	1.85E-05	2.41E-06	2.69E+05
U-238	4.70E+09 y	1.18E+08	1.26E-05	2.04E-06	2.55E+05

DOE undertook an effort to calculate the missing data. In doing so, it assessed the new internal and external dosimetry models being used by EPA to develop Federal Guidance Report No. 13 (Eckerman et al 1998). DOE staff at ORNL had performed similar calculations for the two previous Federal Guidance Reports. When completed, Federal Guidance Report No. 13 will provide coefficients to allow risk from exposures of the public to be estimated directly for radionuclide concentrations in environmental media. These coefficients will not be applicable to exposures of workers and, depending on the dose and dose rate, may not be applicable to exposures during accidents. The interim report does contain data for isotopes of mercury or iodine or their progeny beyond that found in the earlier reports.

Because the Federal Guidance Report No 13 data was not appropriate for this EIS analysis, the ORNL staff developed inhalation and ingestion dose conversion factors for occupational and accident exposure to SNS mercury isotopes with half-lives of more than a few seconds and for SNS iodine isotopes. It also developed factors for immersion and ground plane exposures for the mercury and iodine isotopes (Eckerman 1998b). Dose conversion factors for internal exposures include the contributions of the progeny that are produced by decay in the body following the intake; however, unless the progeny have half-lives similar to or longer than the parent, separate factors are not usually calculated for direct intakes of the progeny. Several of the mercury decay chains do contain progeny with half-lives similar to or longer than the parent. DOE subsequently provided updated factors for mercury and these progeny (Eckerman 1998a).

The dose conversion factors used in this EIS for internal exposures are committed effective dose equivalents. Those used for external exposures are effective dose equivalents. The dose conversion factors listed in Table F-12 were selected from these four sources (Eckerman et al 1998; Eckerman 1998; Eckerman 1998b; Eckerman and Ryman 1993) using the following criteria in the order listed:

Inhalation

1. SNS updated DCFs (Eckerman et al 1998).

   - Mercury assumed to be elemental mercury vapor (Class V) based on EPA Mercury Study Report to Congress (PNL 1982) and DOE analysis of chemical forms emitted (Appendix A).

   - Iodine assumed to be Class F based on DOE analysis of the chemical forms emitted (Appendix A).

   - All others, maximum value for any class (Classes F, M, and S).

2. Federal Guidance Report No. 11.

   - Tritium (H-3) assumed be vapor (Class V).

   - Carbon (C) is maximum of value for organic, monoxide, and dioxide forms of carbon.

   - All others, maximum value (Classes D, W, and Y).

Immersion

1. SNS updated data (Eckerman 1998a).

2. Federal Guidance Report No. 12.

Ground Plane

1. SNS updated data (Eckerman 1998a).

2. Federal Guidance Report No. 12.

Ingestion (not used)

1. SNS updated data (Eckerman et al 1998), maximum value for any uptake factor category (f1).

2. Federal Guidance Report No. 11, maximum value for any uptake category (f1).

The classes referred to in these criteria (F, M, S and D, W, Y) are related to the rate an inhaled radionuclide is cleared from the lungs. Class V is a special class for vapors. The uptake factor (f1) is related to the fraction of the radionuclide transferred to blood in the small intestine. There may be several different uptake factors available for ingested radionuclides. This factor is also applicable to inhalation but has a single value for a given inhalation class.

The radionuclides listed in Table F-12 are all those that could reasonably be expected to be released from the SNS and their progeny. An entry of "0" in Table F-12 indicates that the radionuclide does not emit radiation that results in dose for the indicated exposure. An entry of #N/A indicates that no value was listed in the references used. This does not necessarily mean that the dose conversion factor is unknown. The radionuclide may not be absorbed by the body or may emit radiation that is too weak to travel through air to produce external exposure by immersion or standing on contaminated ground. The noble gas isotope Ar-37 is an example of both of the conditions. Ni-59 and Ni-63 are examples of radionuclides that if absorbed by inhalation or ingestion would cause internal exposure, but emit radiation too weak for external exposures to occur.

Toxic Materials Evaluations

This assessment uses Emergency Response Planning Guidelines (ERPGs) to provide estimates of concentration ranges where one might reasonably expect to observe adverse effects from exposure to toxic substances. The values derived for ERPGs are used for emergency planning purposes and are applicable to most individuals in the general population. The ERPG values are not regulatory exposure guidelines, and they do not incorporate the safety factors normally included in healthy worker exposure guidelines.

The ERPGs were developed by the American Industrial Hygiene Association to aid emergency planners and emergency responders in dealing with hazardous material incidents. The ERPG values are classified in three categories:

ERPG-1	Maximum airborne concentration below which nearly all individuals could be exposed for up to 1 hour without experiencing other than mild transient adverse health effects or perceiving a clearly defined objectionable odor.
ERPG-2	Maximum airborne concentration below which nearly all individuals could be exposed for up to 1 hour without experiencing or developing irreversible or other serious health effects or symptoms that could impair their abilities to take protective action.
ERPG-3	Maximum airborne concentration below which nearly all individuals could be exposed for up to 1 hour without experiencing or developing life-threatening health effects.

In accident conditions at the SNS, the only anticipated hazardous material release is mercury, and mercury is not among the 69 chemicals for which ERPG values have been established. In such a situation, the DOE Emergency Management Advisory Committee, Subcommittee on Consequence Analysis and Protective Actions (SCAPA) have recommended Temporary Emergency Exposure Limits (TEELs). TEELs are interim, temporary or ERPG-equivalent exposure limits for 297 chemicals, including mercury, whose values have not been finalized as ERPGs. The TEEL levels for mercury (elemental and inorganic) adapted by SCAPA in 1996 include:

TEEL-0	0.05 mg/m3
TEEL-1	0.075 mg/m3
TEEL-2	0.1 mg/m3
TEEL-3	10 mg/m3

In this analysis, site-specific meteorology is used to estimate mercury concentrations at the position of the uninvolved worker (within 2000 m of the release point) and the maximum exposed individual of the general public (at the site boundary). The estimated concentrations are then compared to the mercury TEEL values in order to determine the anticipated consequences for comparison between alternative locations.

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